September 10, 2004

Mr. L. M. Stinson
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS TO FACILITATE IMPLEMENTATION OF INDUSTRY INITIATIVE NEI 97-08 “STEAM GENERATOR PROGRAM GUIDELINES” (TAC NOS. MC3667 AND MC3668)

Dear Mr. Stinson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 163 to Facility Operating License No. NPF-2 and Amendment No. 156 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments consists of changes to the Technical Specifications (TSs) in response to your application dated June 28, 2004, as supplemented by letter dated August 5, 2004.

The amendments revise TS 3.4.13, “RCS [Reactor Coolant System] Operational Leakage,” TS 5.59, “Steam Generator [SG] Tube Surveillance Program,” and TS 5.610, “Steam Generator Tube Inspector Report.” They also add a new TS 3.4.17, “Steam Generator Tube Integrity.” These changes facilitate the implementation of industry initiative NEI [Nuclear Energy Institute] 97-08, “Steam Generator Program Guidelines,” which allows a comprehensive, performance-based approach to managing SG performance at Farley Nuclear Plant, Units 1 and 2.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 163 to NPF-2
2. Amendment No. 156 to NPF-8
3. Safety Evaluation

cc w/encl: See next page

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cc w/encl: See next page

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DATE	9/10/04	8/31/04	09/03/04	9/10/04

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JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS TO FACILITATE IMPLEMENTATION OF INDUSTRY INITIATIVE NEI 97-08 “STEAM GENERATOR PROGRAM GUIDELINES” (TAC NOS. MC3667 AND MC3668)

Dated: September 10, 2004

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 163
License No. NPF-2

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated June 28, 2004, as supplemented by letter dated August 5, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 163, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mary Jane Ross-Lee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 10, 2004

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 156
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated June 28, 2004, as supplemented by letter dated August 5, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 156, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mary Jane Ross-Lee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 10, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 163

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

ATTACHMENT TO LICENSE AMENDMENT NO. 156

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
TOC ii	TOC ii
3.4.13-1	3.4.13-1
3.4.13-2	3.4.13-2
—	3.4.17-1
—	3.4.17-2
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	—
5.5-16	—
5.5-17	—
5.5-18	—
5.6-6	5.6-6
B TOC ii	B TOC ii
B TOC iii	B TOC iii
B 3.4.4-2	B 3.4.4-2
B 3.4.5-3	B 3.4.5-3
B 3.4.6-3	B 3.4.6-3
B 3.4.7-3	B 3.4.7-3
B 3.4.13-2	B 3.4.13-2
B 3.4.13-3	B 3.4.13-3
B 3.4.13-4	B 3.4.13-4
B 3.4.13-5	B 3.4.13-5

Remove	Insert
B 3.4.13-6	B 3.4.13-6
----	B 3.4.17-1
----	B 3.4.17-2
----	B 3.4.17-3
----	B 3.4.17-4
----	B 3.4.17-5
----	B 3.4.17-6
----	B 3.4.17-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated June 28, 2004, as supplemented by letter dated August 5, 2004, Southern Nuclear Operating Company (the licensee) submitted a technical specification (TS) amendment request for the Farley Nuclear Plant, Units 1 and 2, concerning the maintaining of steam generator (SG) tube integrity (References 1 and 2). This amendment request is the culmination of NRC and industry efforts since the mid-1990s to develop a programmatic, largely performance-based regulatory framework for ensuring SG tube integrity.

The supplement, dated August 5, 2004, made minor revisions to the proposed tube integrity performance criteria and provided additional information that clarified the amendment request. This revision did not amend the scope of the amendment request as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration as published in the *Federal Register*.

The scope of the Farley TS amendment request includes:

a. Revised Table of Contents
b. Revised TS 3.4.13 and TS Bases B 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE"
c. New TS 3.4.17 and new TS Bases B 3.4.17, "Steam Generator Tube Integrity"
d. Revised TS 5.5.9, "Steam Generator Tube Surveillance Program"
e. Revised TS 5.6.10, "Steam Generator Tube Inspection Report"
f. Revised TS Bases B 3.4.4, "RCS Loops - Modes 1 and 2"
g. Revised TS Bases B 3.4.5, "RCS Loops - Mode 3"
h. Revised TS Bases B 3.4.6, "RCS Loops - Mode 4"
i. Revised TS Bases B 3.4.7, "RCS Loops - Mode 5"

The proposed new TS 3.4.17, in conjunction with the proposed revisions to administrative TS 5.5.9, would establish a new programmatic, largely performance-based framework for ensuring SG tube integrity. TS Bases B 3.4.17 documents the licensee's bases for this framework. Proposed TS 3.4.17 would establish new limiting conditions for operation (LCOs) related to SG tube integrity; namely, (1) SG tube integrity shall be maintained, and (2) all SG tubes satisfying the tube repair criteria (i.e., tubes with measured flaw sizes exceeding the tube repair criteria) shall be plugged in accordance with the SG Program. TS 3.4.17 would include surveillance requirements (SRs) to verify that the above LCOs are met in accordance with the SG Program.

Proposed administrative TS 5.5.9, “Steam Generator Program,” would replace the current administrative TS 5.5.9, ”Steam Generator Tube Surveillance Program.” This revised TS would require establishing and implementing a program that ensures that SG tube integrity is maintained. Tube integrity is defined in the proposed TS in terms of specified performance criteria for structural and leakage integrity. TS 5.5.9 would also provide for monitoring the condition of the tubes relative to these performance criteria during each SG tube inspection and for ensuring that tube integrity is maintained between scheduled inspections of the SG tubes. TS 5.5.9 would retain the current depth-based tube repair limit of 40 percent of the initial tube wall thickness, requiring that tubes satisfying the criteria be plugged.

The proposed changes to TS 5.6.10, “Steam Generator Tube Inspection Report,” revise the existing requirements for, and the contents of, the SG tube inspection report consistent with the proposed revisions to TS 5.5.9. The current requirement for a 12-month report would be changed to a 180-day report.

The Farley amendment package includes proposed revisions to TS 3.4.13 and its bases, “RCS Operational LEAKAGE.” The proposed changes would delete the current LCO limit of 450 gallons per day (gpd) for total primary-to-secondary leakage through all SGs, but would retain the current LCO limit of 150 gpd for primary-to-secondary leakage from any one SG. Retaining this latter requirement effectively ensures that total primary-to-secondary leakage through all the SGs is not allowed to exceed 450 gpd. (Note, Farley, Units 1 and 2, are three-loop plants.) The proposed changes would also revise the TS 3.4.13 conditions and SRs to better clarify the requirements related to primary-to-secondary leakage.

Finally, the TS Bases for TS 3.4.4, 3.4.5, 3.4.6, and 3.4.7 would be revised to eliminate the reference to the Steam Generator Tube Surveillance Program as the method for ensuring SG OPERABILITY.

2.0 REGULATORY EVALUATION

2.1 Current Licensing Basis/SG Tube Integrity

The SG tubes in pressurized-water reactors (PWRs) have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage severe accidents.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 states that the RCPB shall have “an extremely low probability of abnormal leakage...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leak tight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code). Section 50.55a further requires, in part, that throughout the service life of a PWR facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

In the 1970s, Section XI requirements pertaining to ISI of SG tubing were augmented by additional SG tube SRs in the TSs. Paragraph (b)(2)(iii) of 10 CFR, Part 50.55a, states that where TS SRs for SGs differ from those in Article IWB-2000 of Section XI of the ASME Code, the ISI program shall be governed by the TSs.

The existing plant TSs include LCOs and accompanying SRs and action statements pertaining to the integrity of the SG tubing. SG operability in accordance with the SG tube surveillance program is necessary to satisfy the LCOs 3.4.4, 3.4.5, 3.4.6, and 3.4.7 governing RCS loop operability, as stated in the TS Bases B 3.4.4, B 3.4.5, B 3.4.6, and B 3.4.7. In addition, SR 3.4.13.2 accompanying LCO 3.4.13, “RCS Operational LEAKAGE,” requires that tube integrity be verified in accordance with the SG tube surveillance program. The SG tube surveillance program requirements are contained in administrative TS 5.5.9. TS 5.5.9 states that the SGs are to be determined OPERABLE after the actions required by the surveillance program (in Table 5.5.9.2) are completed. LCO 3.4.13 establishes limits on allowable primary-to-secondary LEAKAGE through the SG tubing. Accompanying SRs require verification that RCS operational LEAKAGE is within limits every 72 hours by an RCS water inventory balance (SR 3.4.13.1) and that SG tube integrity is in accordance with the SG tube surveillance program (SR 3.4.13.2).

Under the plant TS SG surveillance program requirements, licensees are required to monitor the condition of the SG tubing and to perform repairs, as necessary. Specifically, licensees are required by the plant TSs to perform periodic ISIs and to remove from service, by plugging, all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit" (old terminology) or "tube repair criteria" (new terminology). The frequency and scope of the inspection and the tube repair limits are specified in the plant TSs.

The plugging limits in the TSs were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III) and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between ISIs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs) such as an SG tube rupture (SGTR) and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR 100 guidelines for offsite doses, GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

2.2 10 CFR 50.36

The licensee is proposing amendments to its plant TSs. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. This regulation requires that the TS include items in five specific categories. These categories include 1) safety limits, limiting safety system settings and limiting control settings; 2) limiting conditions for operation; 3) surveillance requirements; 4) design features; and 5) administrative controls. However, the regulation does not specify a particular TS to be included in a plant's license.

Additionally, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a LCO is required to be included in the TS for a certain item. These criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The NRC staff has reviewed the proposed changes to ensure that these changes conform with 10 CFR 50.36 as discussed herein.

2.3 Background - Technical Specification Amendment Request

The current TS requirements for inspection and repair of SG tubing date to the mid-1970s and define a prescriptive approach for ensuring tube integrity. This prescriptive approach involves inspection of the tubing at specified intervals, implementation of specified tube inspection sampling plans, and repair or removal from service by plugging all tubes found by inspection to contain flaws in excess of specified flaw repair criteria. However, as evidenced by operating experience, the prescriptive approach defined in the TSs is not sufficient in-of-itself to ensure that tube integrity is maintained. For example, in cases of low to moderate levels of degradation, the TSs only require that 3 to 21 percent of the tubes be inspected, irrespective of whether the inspection results indicate that additional tubes may need to be inspected to reasonably ensure that tubes with flaws that may exceed the tube repair criteria, or that may impair tube integrity, are detected. In addition, the TSs (and ASME Code, Section XI) do not explicitly address the inspection methods to be employed for different tube degradation mechanisms or tube locations, nor are the specific objectives to be fulfilled by the selected methods explicitly defined. Also, incremental flaw growth between inspections can, in many instances, exceed what is allowed in the specified tube repair criteria. In such cases, the specified inspection frequencies may not ensure reinspection of a tube before its integrity is impaired. In short, the current TS SRs do not require licensees to actively manage their SG surveillance programs such as to provide reasonable assurance that tube integrity is maintained.

In view of the shortcomings of the current TS requirements, licensees experiencing significant degradation problems have frequently found it necessary to implement measures beyond minimum TS requirements to ensure that adequate tube integrity is being maintained. Until the early 1990s, these measures tended to be ad hoc. In 1994, the NRC staff began work on developing a new regulatory framework to ensure the effectiveness and consistency of licensee programs for ensuring tube integrity. In 1997, this effort took the form of a draft generic letter requesting that licensees upgrade their TSs as necessary to ensure tube integrity, including preparation of accompanying guidance defining an acceptable approach for meeting this objective (Reference 3). The NRC staff developed draft Regulatory Guide (RG) DG-1074, "Steam Generator Tube Integrity" (Reference 4), to support development of a revised regulatory framework. DG-1074 described a performance-based, programmatic approach for ensuring SG tube integrity.

By letter dated December 16, 1997 (Reference 5), the Nuclear Energy Institute (NEI) provided NRC with a copy of NEI 97-06 (Original), "Steam Generator Program Guidelines," and informed the NRC of the following formal industry position.

> Each licensee will evaluate its existing steam generator program and, where necessary, revise and strengthen program attributes to meet the intent of the guidance provided in NEI 97-06, "Steam Generator Program Guidelines," no later than the first refueling outage starting after January 1, 1999.

The stated objectives of this initiative were to have a clear commitment from utility executives to follow industry SG related guidelines developed through Electric Power Research Institute (EPRI), to assure a unified industry approach to emerging SG issues and to apply tube integrity performance criteria in conjunction with the performance-based philosophy of the maintenance rule, 10 CFR 50.65. Reference 6 is the most recent update to NEI 97-06 available to the NRC staff. NEI 97-06 provides general, high-level guidelines for a programmatic, performance-based approach to ensuring SG tube integrity. NEI 97-06 references a number of detailed EPRI guideline documents for programmatic details. NEI 97-06 and its referenced detailed industry guidelines are conceptually similar to DG-1074, but with some important differences with respect to details. Subsequently, the NRC staff elected to discontinue its effort to issue a generic letter while it worked with the industry to resolve NRC staff concerns with the industry initiative and to identify needed changes to the plant TSs to ensure that tube integrity is maintained (References 7 and 8).

Ultimately, in accordance with the performance-based objective of this initiative, the NRC staff determined it was not necessary for the NRC staff to formally review or endorse the NEI 97-06 guidelines or the EPRI guideline documents referenced by NEI 97-06. Instead, the NRC staff has evaluated, as documented herein, proposed changes to the TSs for Farley, Units 1 and 2, which are programmatically consistent with the industry's NEI 97-06 initiative, and which will ensure that the licensee will implement an SG program that provides reasonable assurance that SG tube integrity will be maintained. This TS amendment request constitutes a lead plant submittal based on the industry's NEI Steam Generator Generic License Change Package initiative (Reference 9).

3.0 TECHNICAL EVALUATION

3.1 TS 3.4.17, "Steam Generator (SG) Tube Integrity"

The current TS establishes an operability requirement for the SG tubing; namely, the tubes shall be determined OPERABLE after completion of the actions defined in the SG tube surveillance program (TS 5.5.9). In addition, this surveillance program (and SG operability) is directly invoked by TS 3.4.13 which contains the LCO relating to RCS leakage. However, these TSs do not directly require that tube integrity be maintained. Instead, they require implementation of an SG tube surveillance program which is assumed to ensure tube integrity, but, as discussed above, may not ensure tube integrity depending on the circumstances of degradation at a plant.

To address this shortcoming, the Farley TS amendment package includes a proposed new specification, TS 3.4.17, "Steam Generator (SG) Tube Integrity," which includes a new LCO requirement and accompanying conditions, required actions, completion times, and SRs. The new LCO is applicable in MODES 1, 2, 3, and 4 and requires: 1) SG tube integrity shall be maintained, AND 2) all SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program (specified in the proposed TS 5.5.9). This LCO supplements the LCO in TS 3.4.13 to directly make tube integrity an operating restriction. This is consistent with Criterion 2 of 10 CFR 50.36(c)(2)(ii) since the assumption of tube integrity as an initial condition is implicit in DBA analyses (with the exception of analysis of a design-basis SGTR where one tube is assumed to not have structural integrity) and is acceptable to the NRC staff.

Proposed SR 3.4.17.1 would require that SG tube integrity be verified in accordance with the Steam Generator Program which is described in proposed revisions to TS 5.5.9. The required frequency for this surveillance would also be in accordance with the SG Program, thus meeting the requirements of 10 CFR 50.36(c)(3). The revised TS 5.5.9 would define tube integrity in terms of satisfying tube integrity performance criteria for tube structural integrity and leakage integrity as specified therein. SR 3.4.17.1 would replace the existing surveillance requirement (SR 3.4.13.2) in the RCS Operational LEAKAGE TS (TS 3.4.13), which provides that tube integrity be verified in accordance with the SG surveillance program as provided in the current TS 5.5.9. The proposed SR 3.4.17.1 improves upon the current SR 3.4.13.2 in that it refers to a program that is directly focused on maintaining tube integrity rather than on implementing a prescriptive surveillance program that, as discussed above, may not be sufficient to ensure tube integrity is maintained. Proposed SR 3.4.17.2 would require verification that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the SG Program. The tube repair criteria are contained in the SG Program (proposed TS 5.5.9), and satisfying the tube repair criteria would require that tubes found by ISI to contain flaws with depths equal to or exceeding 40 percent of the nominal tube wall thickness be plugged. The required frequency for SR 3.4.17.2 is prior to entering MODE 4 following a SG tube inspection. The NRC staff concludes that SR 3.4.17.1 and SR 3.4.17.2 are sufficient to determine whether the proposed LCO is met, meet the requirements of 10 CFR 50.36(c)(3), and are acceptable.

The licensee has proposed conditions, required actions, and completion times for the new LCO 3.4.17 as shown in Table 1. The proposed TS 3.4.17 allows separate condition entry for each SG tube.

Table 1 - TS 3.4.17 ACTIONS

Condition	Required Action	Completion Time
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection. <u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection.
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5	6 hours 36 hours

Should SG tube integrity be found by the SG Program to be not maintained (Condition B of TS 3.4.17), Required Actions B.1 and B.2 would require that the plant be in MODE 3 within 6 hours and MODE 5 within 36 hours, respectively. These required actions and completion times are consistent with (1) the general requirements in TS 3.0.3 for failing to meet an LCO and (2) the requirements of TS 3.4.13 when the LCO on primary-to-secondary leakage rate is not met. The NRC staff concludes that these required actions and completion times provide adequate remedial measures should SG tube integrity be found to be not maintained and are acceptable to the NRC staff.

Condition A of proposed TS 3.4.17 addresses the condition where one or more tubes satisfying the tube repair criteria are inadvertently not plugged in accordance with the SG Program. Under Required Action A.1, the licensee would be required to verify within 7 days that tube integrity of the affected tubes is maintained until the next inspection. The accompanying Bases B 3.4.17 states that the tube integrity determination would be based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next inspection. The NRC staff notes that details of how this assessment would be performed are not included in proposed TS 3.4.17 or 5.5.9. The NRC staff finds this to be consistent with having performance-based requirements, finds that the performance criteria (i.e., performance objectives) for assessing tube integrity are clearly defined (in TS 5.5.9), and finds that it is appropriate that the licensee have the flexibility to determine how best to perform this assessment based on what information is and is not available concerning the circumstances of the subject flaw. The proposed 7 days allowed to complete the assessment ensures that the associated risk increment associated with operating with tubes in this condition will be very small. Should the assessment reveal that tube integrity cannot be maintained until the next scheduled inspection or that the assessment is not completed in 7 days, Condition B applies, leading to Required Actions B.1 and B.2, which are evaluated above. Finally, if Required Action A.1 successfully verifies that tube integrity is being maintained until the next inspection, Required Action A.2 would require that the subject tube be plugged in accordance with the SG Program prior to entering MODE 4 after the next refueling outage or SG inspection. Based on the above, the NRC staff concludes that the proposed LCO and accompanying ACTIONS related to failure to plug a tube that satisfies the tube repair criteria to be acceptable.

The licensee has proposed administrative changes to the TS Title page and Bases supporting the proposed new TS 3.4.17. Although the TS Bases are controlled under the auspices of 10 CFR 50.59 and TS 5.5.14, TS Bases Control Program, the NRC staff finds the proposed changes to the proposed TS 3.4.17 Bases to be acceptable.

3.2 Steam Generator Operability

The TS Bases for TS 3.4.4, RCS Loops - MODES 1 and 2; TS 3.4.5, RCS Loops - MODE 3; and TS 3.4.6, RCS Loops - MODE 4; define an OPERABLE RCS Loop as consisting of an OPERABLE reactor coolant pump (RCP) in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the SG Tube Surveillance Program. The Bases for TS 3.4.7, RCS Loops - MODE 5, Loops Filled, define an OPERABLE SG as a SG that can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the SG Tube Surveillance Program. Although the TS Bases are controlled under the auspices of 10 CFR 50.59 and TS 5.5.14, TS Bases Control Program, the licensee has proposed to delete the phrases, “in accordance with the Steam Generator Tube Surveillance Program,” from TS B 3.4.4, B 3.4.5, and B 3.4.6, and “and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program,” from TS B 3.4.7.

With the deletion of these phrases, an OPERABLE SG will be defined under the definition of OPERABLE - OPERABILITY defined in Section 1.1 of the Farley TS and stated below:

> A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

The NRC staff has evaluated the proposed Bases changes. The current Bases refer to the SG Tube Surveillance Program for the requirements of an OPERABLE SG. The SG Tube Surveillance Program provided the controls for the ISI of SG tubes that was intended to ensure that the structural integrity of this portion of the RCS is maintained. Using the definition of OPERABLE - OPERABILITY expands the definition of an OPERABLE SG beyond maintaining structural integrity and is acceptable.

3.3 Proposed Administrative TS 5.5.9, "Steam Generator Program"

The proposed administrative TS 5.5.9, "Steam Generator Program," replaces the existing administrative TS 5.5.9, "Steam Generator Tube Surveillance Program." The current TS 5.5.9 defines a prescriptive strategy for ensuring tube integrity consisting of tube inspections performed at specified intervals, with a specified inspection scope (tube inspection sample sizes), and with a specified tube acceptance limit for degraded tubing, termed tube repair criterion, beyond which the affected tubes must be plugged. The proposed TS 5.5.9 incorporates a largely performance-based strategy for ensuring tube integrity, requiring that a SG Program be established and implemented to ensure tube integrity is maintained. TS 5.5.9 contains only a few details concerning how this is to be accomplished, the intent being that the licensee will have the flexibility to determine the specific strategy to be employed to satisfy the required objective of maintaining tube integrity. However, as evaluated below, the NRC staff concludes that the proposed TS 5.5.9 provides reasonable assurance that the SG Program will maintain tube integrity.

The proposed TS Bases B 3.4.17 states that NEI 97-06 and its referenced EPRI guideline documents will be used to establish the content of the SG Program. The guidelines are industry-controlled documents and licensee SG programs may deviate from these guidelines. Except as may be specifically invoked by the TSs, the NRC staff's evaluation of the Farley amendment package takes no credit for any of the specifics in the guidelines.

3.3.1 Performance Criteria for SG Tube Integrity

Proposed TS 5.5.9 would require that SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage as specified therein.

The NRC staff's criteria for evaluating the acceptability of these performance criteria are that meeting these criteria is sufficient to ensure that tube integrity is within the plant licensing basis and that meeting these criteria, in conjunction with implementation of the SG Program, ensures no significant increase in risk. These performance criteria must also be evaluated in the context of the overall SG Program such that if the performance criteria are inadvertently exceeded, the consequences will be tolerable before the situation is identified and corrected. In addition, the performance criteria must be expressed in terms of parameters that are measurable, directly or indirectly.

3.3.1.1 Structural Integrity Criteria

The proposed structural integrity criteria is as follows:

> All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes maintaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading considerations associated

with design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to differential pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

The NRC staff has evaluated these proposed criteria for consistency with the safety factors embodied in the current licensing basis; specifically, the safety factors embodied in the TS tube repair criteria. The tube repair criterion typically specified in plant TSs is 40 percent of the initial tube wall thickness. This criterion is typically applicable to all tubing flaws found by inspection, except for certain flaw types at certain locations for which less restrictive repair criteria may be applicable (as specified in the TSs) and for certain sleeve repairs for which a more restrictive tube repair criterion may be specified. For Farley, Units 1 and 2, the 40 percent tube repair criterion is the only such criterion and is applicable to all flaw types at all tube locations.

The technical basis by which the NRC staff evaluates tube repair criteria was described in testimony in 1975 by James Knight of the NRC staff before the Atomic Safety and Licensing Board (Reference 10). In addition, in 1976 the NRC staff prepared RG 1.121 (Draft), “Basis for Plugging Degraded PWR Steam Generator Tubes,” (Reference 11) describing a technical basis for the development of tube repair criteria. This draft RG was issued for public comment, but was never finalized. Although not finalized, the RG is generally cited in licensee and industry documentation as the bases for the TS tube repair criteria. The James Knight testimony and draft RG cite the following with respect to safety margins:

a. Degraded tubing should retain a factor of safety against burst of not less than three under normal operating conditions.

b. Degraded tubing should not be stressed beyond the elastic range of the tube material during the full range of normal reactor operation. The draft regulatory guide also states that loadings associated with normal plant conditions, including startup, operation in the power range, hot standby, and cooldown, as well as all anticipated transients (e.g., loss of electrical load, loss of off-site power) that are included in the design specifications for the plant, should not produce a primary membrane stress in excess of the yield stress of the tube material at operating temperature.

c. Degraded tubes should maintain a margin of safety against tube failure under postulated accidents consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the ASME Code. Note, NB-3225 specifies that the rules in Appendix F of Section III may be used for evaluating these loadings.

The “margin of three” criterion stems from Section III of the ASME Code which, in part, limits primary membrane stress under design conditions to one third of ultimate strength. The proposed structural integrity criterion would limit application of the “margin of three” criterion to only those pressure loadings existing during normal full power, steady state operating conditions. Differential pressures under this condition are plant-specific, ranging from 1250 psi to 1500 psi (Reference 12). However, differential pressure loadings can be considerably higher during normal operating transients; for example, ranging to between 1600 psi to 2150 psi during plant heat ups and cool downs (Reference 12). Given a factor of safety equal to three under normal full power conditions, the factor of safety during heat ups and cool downs can be as low as about two.

The NRC staff had proposed in DG-1074 (Reference 4) that the “factor of three” criterion apply to the full range of normal operating loads consistent with the bases of the ASME Code, Section III, stress limits. The industry responded in a white paper (Reference 12) that this was not the intent of the James Knight testimony or draft RG 1.121 and that the proposed NRC staff criterion would lead to tube repair criteria less than the standard 40 percent criterion for many plants. The NRC staff has independently performed calculations which support the industry’s contention that applying the “factor of three” criterion to the full range of normal operating conditions would lead to tube repair criteria more restrictive than the 40 percent criterion which the NRC staff has accepted since the 1970s. The NRC staff concludes that the “factor of three” criterion for application to normal full power, steady state pressure differentials, as proposed by the licensee and the industry, is consistent with the safety margins implicit in existing TS tube repair criteria and, thus, is consistent with the current licensing basis.

Item b above from the James Knight testimony and draft RG 1.121 is often referred to as the “no yield” criterion. From the James Knight testimony, the purpose of this criterion is to prevent permanent deformation of the tube to assure that degradation of the tube will not occur due to mechanical effects of the service condition. This is consistent with ASME Code, Section III, stress limits which serve to limit primary membrane stress to less than yield. The proposed structural integrity criteria does not include this “no yield” criterion. The industry states in its white paper (Reference 12) that if a tube satisfies the “factor of three” criterion at full power operating pressure differentials, then the tube will generally satisfy the “no yield” criterion for the operating transient (e.g., heat up and cool downs) pressure differentials. The white paper acknowledges that this may not be true for all plant-specific conditions and material properties. For this reason, NEI 97-06, Rev. 1, and the EPRI Steam Generator Integrity Assessment Guidelines state that, in addition to meeting the safety factor of three for normal steady state operation, the integrity evaluation shall verify that the primary pressure stresses do not exceed the yield strength for the full range of normal operating conditions. The white paper, which has been incorporated as part of the EPRI Steam Generator Integrity Assessment Guidelines, recommends that this be demonstrated for each plant using plant-specific conditions and material properties.

The NRC staff concurs that the “no yield” criterion need not be specifically spelled out in the TS definition of the structural integrity criteria. The NRC staff finds that the appropriate focus of the TS criteria should be on preventing burst. The NRC staff calculations confirm that the proposed “factor of three” criterion bounds or comes close to bounding the “no yield” criterion for most of the cases investigated. This is not absolute, however. For once-through SGs (OTSGs), the NRC staff noted a case where elastic hoop stress in a uniformly thinned tube could exceed the yield strength by 20 percent under heat up and cool down conditions and still satisfy the “factor of three” criterion against burst under normal steady state, full power operating conditions. Such a tube would still retain a factor of safety of two against burst under heat up and cool down. The amount of plastic strain induced would be limited to between 1 and 2 percent based on typical strain hardening characteristics of the material. This is quite small compared to cold working associated with fabrication of tube u-bends and tube expansions. Operating experience shows that this level of plastic strain (i.e., permanent strain caused by exceeding the yield stress) has not adversely affected the stress corrosion cracking resistance of OTSG tubing relative to that expected for non-plastically strained tubing. Thus, the NRC staff concludes that the "factor of three" criterion is sufficient to limit plastic strains to values that will not contribute significantly to degradation of the tubing and that the "no yield' criterion need not be specifically spelled out in the structural integrity performance criteria.

The proposed safety factor of 1.4 against burst applied to design-basis primary-to-secondary pressure differentials derives from the 0.7 times ultimate strength limit for primary membrane stress in the ASME Code, Appendix F, F-1331.1(a). This criterion is consistent with the stress limit criteria used to develop the standard 40 percent tube repair criteria in the TSs and with the safety factor criteria used in the derivation of alternate tube repair criteria in plant TSs, such as the voltage based criteria for outer-diameter stress corrosion cracking. Thus, the criterion is consistent with the current licensing basis and is acceptable.

Apart from differential pressure loadings, other types of loads may also contribute to burst. Examples of such loads include bending moments on the tubes due to flow induced vibration, earthquake, and loss-of-coolant accident (LOCA) rarefaction waves. For OTSGs, axial loads are induced in the tubes due to pressure loadings acting on the SG shell and tube sheets and due to differential thermal expansion between the tubes and the SG shell. Such non-pressure loads generally produce negligible primary stress during normal operating conditions from the standpoint of influencing burst pressure. In general, such non-pressure loads may be more significant under certain accident loadings depending on SG design, flaw location, and flaw orientation. Such non-pressure sources of primary stress under accident conditions were explicitly considered in the development of the 40 percent tube repair criteria relative to ASME Code, Appendix F, stress limits.

The proposed structural criterion requires that, apart from the safety-factor requirements applying to pressure loads, additional loads associated with DBAs, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine whether these loads contribute significantly to burst or collapse. The NRC staff notes that examples of such additional loads include bending moments during LOCA, MSLB, or safe shutdown earthquake (SSE) and axial, differential thermal loads. "Combination of accidents" refers to the fact that the design and licensing basis for many plants is that DBAs, such as LOCA and MSLB, are assumed to occur concurrently with SSE. Whereas "burst" is the failure mode of interest where primary-to-secondary pressure loads are dominant, "collapse" is a potential limiting failure mode (although an unlikely one, according to industry based on a recent study (Reference 13)) for loads other than pressure loads. "Collapse" refers to the condition where the tube is not capable of resisting further applied loading without unlimited displacement. Although the occurrence of a collapsed tube or tubes would not necessarily lead to perforation of the tube wall, the consequences of tube collapse have not been analyzed and, thus, the NRC staff finds it both appropriate and conservative to ensure there is margin relative to such a condition.

Where non-pressure loads are determined to significantly contribute to burst or collapse, the proposed structural criteria requires that such loads be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and a 1.0 safety factor on axial secondary loads. The 1.2 safety factor for combined primary loads was derived from the ratio of burst or collapse load divided by allowable load from ASME Code for faulted conditions. Burst or collapse load was assumed to be equal to the material flow stress assuming Code minimum yield and ultimate strength values and a flow stress coefficient of 0.5. Allowable load was determined from ASME Code, Section III, Appendix F, F-1331.3.a, which defines an allowable primary membrane plus bending load for service level d (faulted) conditions. The NRC staff finds this 1.2 safety factor acceptable. The proposed 1.0 safety factor for axial secondary loads goes beyond what is required by the design basis in Section III of the ASME Code, since Section III assumes that a one-time application of such a load cannot lead to burst or collapse. However, this is not necessarily the case for tubes with circumferential cracks. The proposed safety factor criterion of 1.0 is conservative for loads which behave as secondary since it ignores the load relaxation effect associated with axial yielding before tube severance (burst) occurs.

Apart from being consistent with the current licensing basis, NRC risk studies have indicated that maintaining the performance criteria safety factors is important to avoiding undue risk, particularly risk associated with severe accident scenarios involving a fully pressurized primary system and depressurized secondary system and where the tubes may heat to temperatures well above design basis values, significantly reducing the strength of the tubes (Reference 14).

Based on the above, the NRC staff finds that the proposed structural performance criteria is consistent with the margins of safety embodied in existing plant licensing bases. Exceeding these criteria should not lead to consequences that are intolerable provided that such conditions are infrequent and that such conditions, when they occur, are promptly detected and corrected so as to ensure that risk is limited. Even if a tube should degrade to the point of rupture under normal operating conditions, such an occurrence is an analyzed condition with reasonable assurance that the radiological consequences will be acceptable. Finally, the structural performance criteria is expressed in terms of parameters which are measurable. Specifically, structural margins can be directly demonstrated through in situ pressure testing or can be calculated from burst prediction models using as input flaw size measurements obtained by ISI. Thus, the NRC staff finds the proposed structural performance criteria to be acceptable.

3.3.1.2 Accident Leakage Criteria

The proposed accident induced leak rate criteria is as follows:

> The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident leakage is not to exceed 1 gpm [gallons per minute] total for all three SGs.

This performance criteria for accident induced leak rate is consistent with leak rates assumed in the licensing basis accident analyses for purposes of demonstrating that the consequences of DBAs meet the guideline limits in 10 CFR 100 for offsite doses, GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits). These criteria do not apply to design basis SGTR accidents for which leakage corresponding to a postulated double ended rupture of a tube is assumed in the analysis. The proposed criteria ensure that from the standpoint of accident induced leakage the plant will be operated within its analyzed condition and is acceptable.

For certain severe accident sequences involving high primary side pressure and a depressurized secondary system, primary-to-secondary leakage may lead to additional heating of the leaking tube than would be the case were it not leaking, thus increasing the potential for failure of that tube and a consequent large early release. The proposed 1.0 gpm limit on total leakage from all three SGs ensures that the potential for accident induced leakage will be maintained at a level which will not increase severe accident risk.

Exceeding these criteria should not lead to consequences which are intolerable provided that such conditions are infrequent and that such conditions, when they occur, are promptly detected and corrected so as to ensure that risk is minimized. It should be noted that the criteria apply to leakage that could potentially be induced by an accident in the unlikely event that such an accident should occur. Finally, the accident leakage performance criteria is expressed in terms of parameters which are measurable, both directly and indirectly. Specifically, structural margins can be directly demonstrated through in situ pressure testing or can be calculated using leakage prediction models using flaw size measurements obtained by ISI as input.

Based on the foregoing, the NRC staff finds the proposed accident leakage performance criteria to be acceptable.

3.3.1.3 Operational Leakage Criterion

Proposed TS 5.5.9 states that the operational leakage performance criterion is specified in LCO 3.4.13, "RCS Operational Leakage." Given the TS LCO limit, a separate performance criterion for operational leakage is unnecessary for ensuring prompt shutdown should the limit be exceeded. However, operational leakage is an indicator of tube integrity performance, though not a direct indicator. It is the only indicator that can be monitored while the plant is operating. Maintaining leakage to within the limit provides added assurance that the structural and accident leakage performance criteria are being met. Thus, the NRC staff believes that inclusion of the TS leakage limit among the set of tube integrity performance criteria is appropriate from the standpoint of completeness and is therefore acceptable.

3.3.2 Condition Monitoring Assessment

Proposed TS 5.5.9 would require that the SG Program include provisions for condition monitoring assessments as follows:

> Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural and accident induced leakage integrity. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

The NRC staff finds that the proposed requirement for condition monitoring assessments addresses an essential element of any performance-based strategy; namely, the need to monitor performance relative to the performance criteria. Confirmation that the tube integrity criteria are met would confirm that the overall programmatic goal of maintaining tube integrity has been met to that point in time. However, failure to meet the tube integrity criteria would be indicative of potential shortcomings in the effectiveness of the licensee's SG Program and the need for corrective actions relative to the program to ensure that tube integrity is maintained in the future. Failure to meet either the structural or accident leakage performance criteria would be reportable pursuant to 10 CFR 50.72 and 50.73 in accordance with guidelines in Reference 15. In addition, the NRC Regional Office would follow up on such an occurrence as appropriate consistent with the NRC Reactor Oversight Program (ROP) (Reference 16) and the risk significance of the occurrence.

TS 5.5.9 would require that condition monitoring be performed at each ISI of the tubing. The NRC staff's evaluation of the proposed frequency of ISI is addressed in Section 3.3.3 of this safety evaluation.

3.3.3 Inservice Inspection

The proposed TS 5.5.9 would require that the SG Program include periodic tube inspections. This proposal includes a new performance-based requirement that the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next inspection. This is a performance-based requirement that complements the requirement for condition monitoring from the standpoint of ensuring tube integrity is maintained. The requirement for condition monitoring is backward looking in that it is intended to confirm that tube integrity has been maintained up to the time the assessment is performed. The ISI requirement, by contrast, is forward looking. It is intended to ensure that tube inspections in conjunction with plugging of tubes are performed such as to ensure that the performance criteria will continue to be met at the next SG inspection. This would be followed again by condition monitoring at the next SG inspection to confirm that the performance criteria were in fact met, and so on.

With respect to scope and methods of inspection, the proposed TS 5.5.9 would also require that the number and portions of tubes inspected and method of inspection be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. Furthermore, an assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

The NRC staff finds that this proposal concerning the scope and methods of inspection includes a number of improvements relative to the current requirement in TS 5.5.9. Currently, TS 5.5.9 requires that tube inspections be conducted from the point of entry on the hot leg side completely around the u-bend to the top support plate on the cold leg side. Thus, the current TS does not require inspection of tubing on the cold leg side up to the uppermost support plate elevation. Operating experience demonstrates that the entire length of tubing is subject to various forms of degradation. The proposed TS 5.5.9 addresses this issue by requiring cold leg as well as hot leg inspections. Also, the proposed requirement clarifies the licensee's obligation under existing TSs and 10 CFR 50, Appendix B, to employ inspection methods capable of detecting flaws of any type that the licensee believes may potentially be present anywhere along the length of the tube based on a degradation assessment.

The proposed TS 5.5.9 specifically excludes the tubesheet welds and the tube ends beyond the welds from the inspection requirements therein. The NRC staff finds this to be consistent with current actual practice and to be acceptable. The tube ends beyond the tube-to-tubesheet welds are not part of the primary pressure boundary.

The proposed TS 5.5.9 would replace current specific requirements pertaining to the number of tubes to be inspected at each inspection, in part, with a requirement that is performance based; that is, the number and portions of tubes inspected (in conjunction with other elements of inspection) shall be such as to ensure that tube integrity is maintained until the next inspection. The current minimum tube sampling requirement for an SG inspection is 3 percent of the SG tubing at the plant. The purpose of this initial sample is to determine whether active degradation is present and whether there is a need to perform additional inspection sampling. Actual industry practice, consistent with NEI 97-06 and the EPRI Examination Guidelines, Rev. 6, typically involves initial inspection samples of at least 20 percent. If moderate numbers of tubes (i.e., category C-2 as defined in the current TS) are found to contain flaws, the current TS requires that an additional 6 to 18 percent of the tubes be inspected. In many cases this requirement is very non-conservative since no consideration is given to whether uninspected tubes may potentially contain flaws which could challenge the tube integrity performance criteria prior to the next inspection. Current industry practice and the industry guidelines involve substantially higher levels of sampling under these circumstances. This practice has been motivated by a desire to minimize forced outages as well as by the requirement to ensure tube integrity. The NRC staff finds, therefore, that current TS sampling requirements do not drive actual sampling programs in the field for plants with low to moderate levels of tube degradation, and that for moderate levels of tube degradation the current TS requirements do not ensure adequate levels of sampling to ensure tube integrity will be maintained. The proposed TS 5.5.9 addresses this shortcoming by requiring that inspection scope be consistent with the overall performance objective that tube integrity be maintained until the next SG inspection.

For SGs with high levels of degradation (i.e., category C-3 as defined in current TS), the current TS requires that the inspections be expanded to include 100 percent of the tubes in the affected SG. This requirement is conservative in cases where the active degradation is confined to specific groups of tubes in the SG. This requirement does drive actual sampling programs in the field since industry guidelines would permit 100 percent sampling to be confined to those portions of the SG bounding the region where the degradation has been found to be active. The proposed TS 5.5.9 would give licensees the flexibility to implement less than 100 percent inspection of the SG in these cases provided it is consistent with the performance-based objective of ensuring that tube integrity is maintained until the next SG inspection.

Overall, the NRC staff concludes that the proposed TS 5.5.9 ensures that licensees will implement inspection scopes consistent with the overall objective that tube integrity be maintained. To meet this requirement, licensees will find it necessary to inspect tubes that may potentially contain flaws that may challenge the tube integrity performance criteria prior to the next inspection. The proposed TS 5.5.9 gives the licensee the flexibility to define an inspection scope that ensures that this objective is met while avoiding any unnecessary inspections.

With respect to frequency of inspection, TS 5.5.9 currently requires that SG inspections be performed every 24 calendar months. This frequency may be extended to once every 40 calendar months if the previous two inspections revealed only low-level degradation (i.e., category C-1 results as defined in the TS). The inspection frequency is required to revert from the 40 calendar months to 20 calendar months if an extensive level of degradation (i.e., category C-3 results as defined in the TS) is observed during the most recent inspection. Except in cases where extensive degradation (i.e., category C-3) is found in any SG, SGs may be inspected on a rotating basis at each inspection. Thus, for 4-loop plants performing SG inspections at 24-month intervals, intervals for individual SGs may range to 96 months. Similarly, for 4-loop plants performing SG inspections at 40-month intervals, intervals for individual SGs may range to 160 months. However, these prescriptive requirements bear no direct relationship to the overall objective of ensuring that tube integrity is maintained. These requirements apply irrespective of the flaw detection and sizing performance of the inspection methods utilized and the rate at which flaws may be growing in the subject SGs. These requirements do not ensure that flawed tubing remaining in service following an SG tube inspection and the incremental flaw growth that may take place prior to the next inspection are within the allowances provided for by the TS tube repair limit or that tube integrity will be maintained prior to the next inspection.

Plants operating with their originally installed SGs have typically inspected each SG at each refueling outage which typically occur at intervals of less than 24 calendar months. The vast majority of these SGs contained alloy 600 mill annealed (MA) tubing which quickly became moderately to extensively degraded (i.e., category C-2 or C-3 as defined in the TS) such that the TS would not allow longer intervals. The 24-month inspection interval requirement usually proved sufficient in maintaining tube integrity. Nonetheless, there have been instances where licensees have performed mid-cycle inspections to ensure tube integrity would be maintained.

However, many SGs with alloy 600 MA tubing have been replaced with SGs with alloy 600 TT or alloy 690 TT tubing which have proven to be much more resistant to SCC than alloy 600 MA tubing. This includes Farley, Units 1 and 2, which have replacement SGs using alloy 690TT tubing. Based on early low levels of degradation, some of the plants with replacement SGs are taking advantage of the longer inspection intervals permitted by the TS. The NRC staff approved a TS amendment for Farley, Units 1 and 2, authorizing implementation of 40-month inspection intervals immediately following the first ISI of the replacement SGs.

Under the proposed TS 5.5.9, the required frequency of inspection in conjunction with inspection scope and inspection methods shall be such as to ensure that tube integrity is maintained until the next SG inspection. This addresses existing shortcomings in the current requirements in that it requires that inspection frequency be part of a management strategy aimed at ensuring tube integrity. The proposed TS 3.4.17 Bases states that inspection frequency will be determined, in part, by operational assessments which utilize additional information on existing degradation and flaw growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next SG inspection.

The NRC staff also notes, however, that any assessment or projection of the future condition of the SG tubing based on the existing condition of the tubing and anticipated flaw growth rates can involve significant uncertainty that may be difficult to conservatively and reliably bound. For this reason, proposed TS 5.5.9 supplements the performance-based requirement concerning inspection frequencies with a set of prescriptive requirements that provide added assurance that tube integrity will be maintained.

The proposed prescriptive requirements include a requirement that 100 percent of the tubes in each SG be inspected at the first refueling outage following SG replacement. The NRC staff notes that this requirement is a moot point for Farley, Units 1 and 2, for which the first ISI of the replacement SGs has already been performed. The required scope of this inspection is substantially more restrictive than the current requirement which requires a 3 percent sample of the total SG tube population and requires inspection of only two of the three SGs. For Farley, Units 1 and 2, with alloy 690 TT tubing, the proposed TS 5.5.9 would require that 100 percent of the tubes be inspected at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months (EFPM), with the first sequential period being considered to begin at the time of the first ISI of the SGs following SG replacement. This sliding scale is intended to address the increased potential for the initiation of stress corrosion cracking over time. In addition, the licensee would be required to inspect 50 percent of the tubes by the refueling outage nearest the mid-point of the period and the remaining 50 percent by the refueling outage nearest the end of the period. However, no SG shall operate for more than 72 EFPM or three refueling outages (whichever is less) without being inspected. If crack indications are found in any tube, then the next inspection for each SG for the degradation mechanism causing the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less). As a point of clarification, the proposed requirements stipulate that if definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not a crack, then the indication need not be treated as such.

These proposed prescriptive requirements in total, for Farley, Units 1 and 2, cannot be described simplistically as being more restrictive or less restrictive than current requirements. They are a quite different set of requirements, being generally more restrictive for SGs with low-to-moderate levels of degradation (i.e., categories C-1 to C-2 as defined in current TS) to somewhat less restrictive for plants with extensive levels of degradation other than cracks. As previously noted, management of SCC mechanisms relative to the performance criteria poses a particular challenge compared to other degradation mechanisms. The proposed requirement to limit inspection intervals to one refueling outage to address any cracking mechanism found to be present in the SGs is a substantially more restrictive requirement than current TS requirements that apply for plants with low to moderate levels of cracked tubes and for practical purposes leads to the same inspection frequency (every refueling outage) as would be required under current TS requirements for plants with moderate to extensive levels of cracked tubes.

The proposed prescriptive requirements relating to inspection frequency have been developed based on qualitative engineering considerations and experience, reflecting the improved SCC resistance of alloy 690 TT tubing relative to alloy 600 TT and particularly relative to alloy 600 MA tubing, that the potential for cracking increases with increasing time in service, and the particular challenges associated with the management of SCC with respect to satisfying the tube integrity performance criteria. The proposed prescriptive requirements are intended primarily to supplement the performance-based requirement that inspection frequency in conjunction with inspection scope and methods be such as to ensure tube integrity is maintained. This performance-based requirement must be satisfied in addition to the prescriptive requirements. The NRC staff concludes that the proposed performance-based requirement, in conjunction with the proposed prescriptive requirements, represents a significantly more effective strategy for ensuring tube integrity than that provided by current TS requirements and will serve to ensure that tube integrity is maintained between SG inspections.

3.3.4 Tube Repair Criteria

Proposed TS 5.5.9 would retain the current TS tube repair criterion (termed plugging limits in current TSs) requirements. Specifically, TS 5.5.9 would require that tubes found by ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness be plugged. This criterion is fully consistent with the tube integrity performance criteria in that flaws not exceeding the tube repair criterion satisfy the performance criteria. In addition, the criterion provides an allowance for flaw size measurement error and incremental crack growth between inspections. The repair criteria requirement is prescriptive rather than performance-based. It provides added assurance that tube integrity will be maintained, given the performance-based strategy that is also to be followed under the proposed TS 5.5.9. Its inclusion as part TS 5.5.9 also ensures that the NRC staff has the opportunity to review any risk implications should the licensee propose a license amendment for alternate tube repair criteria, in conjunction with alternate tube integrity performance criteria, at some time in the future.

3.3.5 Monitoring of Operational Primary-to-secondary Leakage

Proposed TS 5.5.9 would require that the SG Program include provisions for monitoring primary-to-secondary leakage. The NRC staff's evaluation of this proposal is included as part of the NRC staff's evaluation of the proposed change to TS 3.4.13, "RCS Operational Leakage," in Section 3.5 of this SE.

3.4 <u>TS 5.6.10, "Steam Generator (SG) Tube Inspection Report"</u>

The Farley TS amendment package would revise the reporting requirements of TS 5.6.10. Currently, TS 5.6.10 requires that the complete results of the SG Tube Surveillance Program (i.e., the ISI results) be reported within 12 months following completion of the program and include (1) the number and extent of the tubes inspected, (2) the location and percent of wall thickness penetration for each indication, and (3) identification of tubes plugged. Under the revised requirement, a report shall be submitted within 180 days of entry into MODE 4 following a SG inspection. The report shall include:

- The scope of the inspections performed in each SG,
- active degradation mechanisms found,
- non-destructive examination techniques used for each degradation mechanism,
- location, orientation (if linear), and measured sizes (if available) of service induced indications,
- number of tubes plugged during the inspection outage for each active degradation mechanism,
- total number of tubes plugged during the inspection outage for each active degradation mechanism,
- total number and percentage of tubes plugged to date, and
- the results of condition monitoring, including the results of tube pulls and in-situ testing.

This revised reporting requirement is a more comprehensive requirement than the current 12-month report and will enhance the NRC staff's ability to monitor the kinds of inspections being performed, the extent and severity of each active degradation mechanism, degradation trends (stable or getting worse), and the degree of challenge faced by the licensee in maintaining tube integrity. The 180-day reporting requirement is adequate given that should the SG program fail to maintain tube integrity as indicated by condition monitoring, this would be promptly reportable in accordance with 10 CFR 50.72 and Reference 15 allowing the NRC staff to engage in any follow-up activities which it determines to be necessary.

TS 5.6.10 also currently requires that the numbers of tubes plugged in each SG be reported to the NRC within 15 days following completion of the program. In addition, TS 5.6.10 requires that inspection results falling into Category C-3 shall be reported to the NRC pursuant to 10 CFR 50.73 prior to the resumption of plant operation and that the report include a description of the tube degradation and corrective measures taken to prevent recurrence. The Farley TS amendment package would delete both of these requirements. The NRC staff finds deletion of these requirements to be acceptable. Neither the number of tubes plugged nor the finding of Category C-3 results (i.e., 10 percent of the tubes inspected contain degradation or 1 percent of the tubes inspected satisfy the tube repair criteria) have any real bearing on whether tube integrity is being maintained. The NRC staff also notes that the Farley TS amendment would delete the definition of inspection results categories in the revised TSs. If the SG program is effectively maintaining tube integrity, tubes found to be degraded or to be pluggable will also satisfy the tube integrity performance criteria. The regulation at 10 CFR 50.72, in conjunction with Reference 15, require that the NRC staff be promptly notified in the event that the tube integrity performance criteria are not met. The NRC staff would have the opportunity under the NRC ROP to follow up on such an occurrence as warranted. The regulation at 10 CFR 50.73 requires that a Licensee Event Report be issued within 60 days of the of the finding which addresses, in part, the degraded condition of the tube(s) and corrective measures being taken.

Based on the foregoing, the NRC staff finds the proposed revisions to the reporting requirements in TS 5.6.10 to be acceptable.

3.5 <u>TS 3.4.13, RCS Operational Leakage</u>

The licensee proposed several changes to the LCO, required actions, and SRs for TS 3.4.13, “RCS Operational Leakage.” These changes include administrative changes to the LCO, required action statements, and SR. The proposed administrative changes included the following:

a) adding “and” to the end of LCO 3.4.13.c;

b) replacing “SG” in LCO 3.4.13.e with “steam generator (SG);”

c) LCO 3.4.13.e is changed to LCO 3.4.13.d with the deletion of the existing LCO 3.4.13.d discussed below.

d) adding “operational” to “RCS operational LEAKAGE” in Condition A;

e) adding “or primary-to-secondary LEAKAGE” to the end of Condition A. Condition A will state “RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary-to-secondary LEAKAGE.”

f) modifying the NOTE associated with SR 3.4.13.1. “NOTE” will be changed to “NOTES,” a “1.” and a second note, Note 2, will be added which will state “Not applicable to primary-to-secondary LEAKAGE.”

g) SR 3.4.13.1 Note 1 will be reworded to state “Not required to be performed in MODE 3 or 4 until 12 hours after establishment of steady state operation.”

The NRC staff has reviewed these administrative changes and finds them acceptable. In particular, the addition of “or primary-to-secondary LEAKAGE” to Condition A and SR 3.4.13.1 Note 2 are considered to be administrative changes because these changes support the more restrictive addition of primary-to-secondary LEAKAGE to Condition B and SR 3.4.13.2. The need for Note 2 with respect to SR 3.4.13.1 (i.e., not applicable to primary-to-secondary LEAKAGE) and for the proposed new SR 3.4.13.2, which deals with primary-to-secondary LEAKAGE, is discussed in the proposed revision to the BASES in B 3.4.13.2. The revised BASES states that SR 3.4.13.1 is not applicable to primary-to-secondary leakage because leakage rates of 150 gpd or less cannot be accurately measured by an RCS water inventory balance.

3.5.1 Deletion of LCO 3.4.13.d

LCO 3.4.13.d currently specifies that total primary-to-secondary LEAKAGE through all SGs be limited to 450 gpd and LCO 3.4.13.e specifies that primary-to-secondary LEAKAGE through any one SG be limited to 150 gpd. The licensee states that the 450 gpd limit for LEAKAGE through all SGs is redundant with the 150 gpd limit through any one SG (each Farley unit has three SGs, so 3 x 150 = 450 gpd total leakage through all SGs) and, thus, the licensee is proposing deletion of the 450 gpd limit. Accordingly, the Farley TS amendment package would delete LCO 3.4.13.d, but would retain the 150 gpd limit for any one SG in LCO 3.4.13.e. This revised requirement would allow total LEAKAGE through all SGs to be equal to 450 gpd, assuming all SGs are leaking at the rate of 150 gpd. Because the existing LCO 3.4.13.d is redundant to LCO 3.4.13.e, the NRC staff concludes that deleting LCO 3.4.13.d results in no change to the existing limits on total primary-to-secondary leakage from all SGs. Thus, the NRC staff finds the proposed change to the LCO requirement to be acceptable.

3.5.2 TS 3.4.13 Condition B Primary-to-secondary LEAKAGE

The primary-to-secondary leakage limit provides assurance against tube rupture at normal operating and faulted conditions. This, together with the allowable accident induced leakage limit, helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 100 and GDC 19 dose limits or other NRC-approved licensing basis for postulated faulted events. The licensee proposed to add an additional OR statement to Condition B with regards to primary-to-secondary LEAKAGE. As proposed, Condition B would state:

> “Required Action and associated Completion Time of Condition A not met.
>
> OR
>
> Pressure boundary LEAKAGE exists.

OR

Primary-to-secondary LEAKAGE not within limit."

The current requirements, Condition A, have a completion time of four hours to reduce LEAKAGE (other than pressure boundary LEAKAGE) to within limits after which Condition B (plant shutdown) must be entered. The TS limit is more restrictive than the current requirements in that if primary-to-secondary leakage exceeds 150 gpd, then a plant shutdown must be commenced without an allowance to reduce leakage, as provided in Condition A. The revised Condition B would require the reactor to be in MODE 3 in 6 hours and MODE 5 in 36 hours if primary-to-secondary leakage is not within limits. As discussed in Section 3.5 above, the licensee has excluded primary-to-secondary leakage from Condition A. The NRC staff has reviewed the proposed change to Condition B. These changes are additional restrictions on plant operations that enhance safety; therefore, the NRC staff has concluded that the addition of the primary-to-secondary leakage OR statement to Condition B is acceptable.

3.5.3 Surveillance Requirements - Primary-To-Secondary Leakage

SR 3.4.13.1 currently requires verification that RCS operational LEAKAGE is within limits by performance of RCS water inventory balance. The accompanying BASES, B 3.4.13, states that primary-to-secondary leakage is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems. This BASES further states that the RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. As previously discussed in Section 3.5 of this SE, the licensee has proposed adding a note to SR 3.4.13.1 stating that this particular surveillance requirement is not applicable to primary-to-secondary leakage. The licensee would revise the accompanying BASES, B 3.4.13, justifying this change; namely, LEAKAGE of 150 gpd cannot be measured accurately by an RCS water inventory balance. The licensee has proposed a new SR, SR 3.4.13.2, which would verify with a frequency of 72 hours that primary-to-secondary leakage does not exceed the 150 gpd LCO limit. The NRC staff believes this to be acceptable and in accordance with 10 CFR 50.36(c)(3). The revised requirement would not specify the specific method to be employed; however, proposed TS 5.5.9 would require that the SG Program include provisions for monitoring primary-to-secondary leakage. There are a variety of methods that can be used, and the NRC staff concludes there is no need to tie this surveillance to a specific method in order to ensure that the plant is operated safely and within its LCO limits. The licensee would state in the accompanying BASES, B 3.4.13, that the primary-to-secondary leakage measurement uses continuous process radiation monitors or radio chemical grab sampling. The NRC staff notes that the EPRI PWR Primary-to-Secondary Leak Guidelines provide extensive guidance to this effect.

The revised BASES, B 3.4.13, would also state that primary-to-secondary LEAKAGE is measured against the 150 gpd limit under room temperature conditions as described in the EPRI PWR Primary-to-secondary Leak Guidelines. The BASES in B 3.4.13 state that steamline break (SLB) is the most limiting accident or transient from the standpoint of dose releases from primary-to-secondary LEAKAGE. The Farley safety analysis for SLB assumes 500 gpd and 470 gpd primary-to-secondary LEAKAGE (for room temperature conditions) in the faulted and intact SGs respectively as an initial condition. Thus, the assumed total primary-to-secondary LEAKAGE from all SGs is 1440 gpd (1 gpm). The NRC staff concludes that measurement of operational primary-to-secondary LEAKAGE under room temperature conditions relative to the 150 gpd operational limit is acceptable since it ensures that LEAKAGE under hot operational conditions will be less than assumed in the Farley safety analysis and, thus, is in accordance with 10 CFR 50.36(c)(ii).

The new SR, SR 3.4.13.2, with respect to primary-to-secondary leakage replaces the current SR 3.4.13.2 which involved verifying SG tube integrity in accordance with the SG Tube Surveillance Program. As discussed earlier in this SE, TS 5.5.9, “Steam Generator Tube Surveillance Program,” would be replaced by TS 5.5.9, “Steam Generator Program.” The SR to verify tube integrity would be addressed in the proposed new TS 3.4.17, “Steam Generator Tube Integrity,” SRs.

Based on the above, the NRC staff concludes that the proposed revisions to SR 3.4.13.1 and SR 3.4.13.2 are in accordance with 10 CFR 50.36(c)(3) and 10 CFR 50.36(c)(ii) and are acceptable.

3.6 Technical Evaluation - Summary and Conclusions

The Farley TS amendment package establishes a programmatic, largely performance-based regulatory framework for ensuring SG tube integrity is maintained. The NRC staff finds that it addresses key shortcomings of the current framework by ensuring that SG programs are focused on accomplishing the overall objective of maintaining tube integrity. It incorporates performance criteria for evaluating tube integrity that the NRC staff finds are consistent with the structural margins and the degree of leak tightness assumed in the current plant licensing basis. The NRC staff finds that maintaining these performance criteria provides reasonable assurance that the SGs can be operated safely without increase in risk.

The revised TSs would contain limited details concerning how the SG Program is to achieve the required objective of maintaining tube integrity, the intent being that the licensee will have the flexibility to determine the specific strategy for meeting this objective. However, the NRC staff finds that the revised TSs include sufficient regulatory constraints on the establishment and implementation of the SG Program such as to provide reasonable assurance that tube integrity will be maintained.

Failure to meet the performance criteria will be reportable pursuant to 10 CFR 50.72 and 50.73. The NRC ROP provides a process by which the NRC staff can verify that the licensee has identified any SG Program deficiencies that may have contributed to such an occurrence and that appropriate corrective actions have been implemented.

In conclusion, the NRC staff finds that the Farley TS amendment request conforms to the requirements of 10 CFR 50.36 and establishes a TS framework that will provide reasonable assurance that tube integrity is maintained without undue risk to public health and safety.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The amendment has been evaluated against the three standards in 10 CFR 50.92(c). In its analysis of the issue of no significant hazards consideration, as required by 10 CFR 50.91(a), the licensee has provided the following:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

The structural integrity performance criterion is:

"All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads."

The accident induced leakage performance criterion is:

"The primary to secondary accident induced leakage rate for all design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. For FNP Units 1 and 2, leakage is not to exceed 1 gpm [gallons per minute] total for all three SGs. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved alternate repair criteria."

The operational LEAKAGE performance criterion is:

The RCS operational primary to secondary LEAKAGE through any one SG shall be limited to 150 gpd [gallons per day].

A steam generator tube rupture (SGTR) event is one of the design basis accidents analyzed as part of the plant licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). For FNP Units 1 and 2, these analyses assume that primary to secondary LEAKAGE for all SGs is 1 gpm. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed in this change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change to the TS. The program, defined by NEI 97-06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, plugging, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I-131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the TS for operational leakage and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition. The analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gpm with no more than 500 gpd in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I-131 are at the technical specification values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TS and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TS.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of a MSLB, rod ejection, or a reactor coolant pump locked rotor event.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed performance based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   Response: No

   The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

   Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

   For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

The NRC staff has reviewed the licensee's analysis, and based on this review, has determined that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff finds that the amendment request involves no significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 [and change the surveillance requirements]. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (69 FR 46590). Additionally, the Commission has made a final no significant hazards consideration with respect to this amendment. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter, L. M. Stinson, Vice President, Southern Nuclear Operating Company, to NRC, "Joseph M. Farley Nuclear Plant - Request for Technical Specifications Change - Steam Generator Program," June 28, 2004. ADAMS Accession No. ML041820087.

2. Letter, L. M. Stinson, Vice President, Southern Nuclear Operating Company, to NRC, "Joseph M. Farley Nuclear Plant - Request for Technical Specifications Change - Steam Generator Program - Revision 1," August 5, 2004.

3. COMSECY-97-013, "Steam Generator Rulemaking," May 23, 1997.

4. Federal Register: January 20, 1999, Volume 64, Number 12, Request for Public Comment on Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," December 1998.

5. Letter, R.E. Beedle, NEI, to L. J. Callan, NRC, December 16, 1997, transmitting NEI 97-06 (Original), "Steam Generator Program Guidelines."

6. NEI 97-06, Revision 1, "Steam Generator Program Guidelines," January 2001. ADAMS Accession No. ML010430054.

7. SECY-98-248, “Proposed Generic Letter 98-XX, “Steam Generator Tube Integrity,”“ October 28, 1998.

8. SECY-00-0078, “Status and Plans for Revising the Steam Generator Tube Integrity Regulatory Framework,” March 30, 2000.

9. Letter, NEI Technical Specification Task Force to William D. Beckner, NRC, “TSTF-449, Revision 1,” dated September 9, 2003. ADAMS Accession No. ML032590633.

10. Testimony of James Knight before the Atomic Safety and Licensing Board, January 1975.

11. Draft Regulatory guide 1.121, “Bases for Plugging Degraded PWR Steam Generator tubes,” August 1976.

12. Memorandum dated September 8, 1999, to W. H. Bateman, Chief, EMCB, NRR, NRC from J. W. Anderson, EMCB, NRR, NRC, “Summary of August 27, 1999, Senior Management Meeting with NEI/EPRI/Industry to Discuss Issues Involving Implementation Of NEI 97-06.” This memorandum encloses Industry White Paper entitled, “Deterministic Structural Performance Criterion Pressure Loading Definition.”

13. Memorandum dated May 19, 2004, from J. L. Birmingham, Project Manager, NRR, NRC to Cathy Haney, Program Director, Policy and Rulemaking Program, Division of Regulatory Improvement Programs, NRR, NRC, “Summary of May 14, 2004 Meeting with Nuclear Energy Institute (NEI) on Status of Steam Generator Structural Integrity Performance Criteria.” ADAMS Accession No. ML041540500.

14. NUREG-1570, “Risk Assessment of Severe Accident -Induced Steam Generator Tube Rupture,” March 1998.

15. NUREG-1022, Rev 2, “Event Reporting Guidelines 10 CFR 50.72 and 50.73,” October 31, 2000[1].

16. NUREG-1649, Rev 3, “Reactor Oversight Process,” July 2000.

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[1]On February 18, 2004, a Federal Register notice (69 FR 7661) was issued requesting comments on the NRC’s intent to issue an errata to Revision 2 of NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." The errata would indicate that steam generator tube degradation is considered serious if either of the two criteria specified in Section 3.2.4(A)(3) of NUREG-1022 (i.e., the structural and accident leakage performance criteria), Revision 2, are not satisfied.

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